Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

November 21, 2011

L-2011-488
10 CFR 50.73

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Re: St. Lucie Unit 2
Docket No. 50-389
Reportable Event: 2011-002-01
Date of Event: June 6, 2011
Unplanned Automatic Reactor Trip During Reactor Protection System Testing - Supplement

The attached Licensee Event Report LER 2011-002 supplement is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(iv)(A). Added or revised text is marked with revision bars.

This supplement supersedes in its entirety the previously submitted LER 2011-002 by Florida Power & Light (FPL) letter L-2011-279, dated August 5, 2011. The Corrective actions and Analysis of the Safety Significance were not changed as a result of the supplement.

If there are any questions, please call Eric Katzman, Licensing Manager, at (772) 467-7734.

Very truly yours,

Richard L. Anderson
Site Vice President
St. Lucie Plant

RLA/dlc
Attachment

IE22
NRR

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resourse@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

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4. TITLE

Unplanned Automatic Reactor Trip During Reactor Protection System Testing

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
06	06	2011	2011	- 002 -	01	11	21	2011	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE

1

10. POWER LEVEL

100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Don Cecchett	772-467-7155

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
A	JC	94	G080	YES					

14. SUPPLEMENTAL REPORT EXPECTED

☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On June 6, 2011 St. Lucie Unit 2 was in Mode 1, when an unplanned automatic reactor trip occurred. The trip occurred while Operators were performing a reactor protection system (RPS) test in which the Operator inadvertently moved the matrix relay trip select switch one position too far, causing the reactor trip circuit breakers (TCBs) to open. Upon reactor trip all CEAs fully inserted into the core, auxiliary feedwater actuation system (AFAS) initiated as designed on low steam generator water level and was restored using auxiliary feedwater. All safety related systems functioned as designed.

A root cause evaluation (RCE) determined the trip was a result of latent single human error vulnerability in the test methodology. Contributing causes included human error during performance of the test and inadequate "problem resolution" which required the matrix relay hold pushbutton to be depressed during performance of the entire test. This event is reportable under 10 CFR 50.73(a)(2)(iv)(A), as any event or condition that resulted in a manual or automatic reactor trip and actuation of the auxiliary feedwater system.

Corrective actions include procedure revisions to test methodology to remove the human error vulnerability and replacement of the RPS matrix relay hold pushbuttons with rotary switches.

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NARRATIVE

Description of the Event

On June 6, 2011 while performing RPS [EIIS:JC] Logic Matrix testing per plant procedure 2-OPS-63.01, step 4.10.3.K directed that the matrix relay trip select switch be placed at mid-position between position 2 and position 3 to facilitate resetting and closing of two trip circuit breakers when the operator attempted to operate the switch his hand slipped off the switch. A second attempt to position the switch was made which resulted in the switch moving from position 2, past the mid-position and on to position 3, causing a reactor trip. Upon reactor trip all CEAs fully inserted into the core, auxiliary feedwater actuation system (AFAS) initiated as designed on low steam generator water level and was restored using auxiliary feedwater. All safety related systems functioned as designed.

Cause

A root cause determined the event was "human error vulnerability" resulting from a previous 1998 procedure change in the test methodology which required depressing the matrix relay hold pushbutton during the performance of the entire test, placing the circuit in a ready-to-actuate state.

A contributing cause was human error on the part of the operator during performance of the two-out-of-four trip functional test.

Inadequate "problem resolution" was also identified as a contributing cause. In 1998 a problem during the test was corrected by adding a caution note to the test procedure to maintain the matrix relay hold pushbutton depressed during the entire test. Additionally, use of a mechanical aid to keep the pushbutton depressed was approved without adequate benchmarking of Combustion Engineering plants with a similar RPS design.

Analysis of the Event

This event is reportable under 10 CFR 50.73(a)(2)(iv)(A), as any event or condition that resulted in a manual or automatic activation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section. The systems listed under (a)(2)(iv)(B) are (1) reactor trip and (6) auxiliary feedwater system.

Analysis of Safety Significance

This event is considered an uncomplicated event, and although not specifically analyzed in the Chapter 15 accident analyses it is bounded by Chapter 15 analyzed events. Analyzed events are categorized into four categories. These categories are defined as Condition I, Normal operation and Operational transients, Condition II, Faults of moderate frequency, Condition III, Infrequent faults and Condition IV Limiting faults. The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. This event would correlate to a Condition II event. ANS Condition II occurrences are faults that may occur with moderate frequency during the life of the plant. They are accommodated with, at most, a reactor shutdown with the plant being capable of returning to operation after a corrective action. In addition, no ANS Condition II

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NARRATIVE

occurrence shall cause consequential loss of function of fuel cladding and reactor coolant system barriers.

Since all safety related systems functioned as designed there is no impact on the health and safety of the public.

The conditional core damage probability (CCDP) and conditional large early release probability (CLERP) associated with having an automatic reactor trip due to human action over 24 hrs is 1.17E-09 and 1.0E-12, respectively. Based on these values, the safety significance due to this event is minimal (Green).

Corrective Actions

The corrective actions and supporting actions listed below are entered into the site corrective action program. Any changes to the actions will be managed under the corrective action program.

1. Revise procedure 1/2-OSP-63.01 to remove the requirement to hold the Matrix relay hold pushbutton pressed in during the entire two-out-of-four trip functional testing sections.
2. Implement Engineering Change to replace the matrix relay hold pushbuttons for RPS and AFAS with rotary switches for both Unit 1 and 2.
3. Change the monthly functional test procedure for AFAS to reposition the system select switch to the off position prior to depressing the Matrix Relay Hold Pushbutton.

Similar Events

A search of the correction action (CR) database for the last five years revealed no automatic reactor trips due to human error.

Industry operating experience from other Combustion Engineering Plants could have possibly avoided this event.

Failed Components

NA